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Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

August 2, 2007

NRC 2007-0064
10 CFR 50.73

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Point Beach Nuclear Plant Units 1
Docket 50-266
Renewed License No. DPR-24

Licensee Event Report 266-2007-004-00
Manual Reactor Trip Due To Loss of Feedwater Regulating Valve Control

Enclosed is Licensee Event Report 266-2007-004-00 for Point Beach Nuclear Plant Unit 1. This LER discusses the manual reactor trip initiated in response to a main feedwater regulating valve feedback arm-to-positioner linkage separation. This event is reportable in accordance with 10 CFR 50.73(a)(2)(iv)(A) for, "Any event or condition that resulted in manual actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of this section."

This letter contains no new commitments and no revisions to existing commitments.

Dennis L. Koehl

Dennis L. Koehl
Site Vice-President, Point Beach Nuclear Plant
Nuclear Management Company, LLC

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
PSCW

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 6-30-2007

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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TITLE (4)

Manual Reactor Trip Due To Loss of Feedwater Regulating Valve Control

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
06	05	2007	2007	-- 004 --	00	08	02	2007	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1
POWER LEVEL (10)	100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR •: (Check all that apply) (11)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Tom Staskal, Compliance Engineer	920-755-7621

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).		NO				

ABSTRACT

On June 5, 2007, Unit 1 was operating in Mode 1 at 100% power. At 1512 hours water level in 'B' steam generator was observed to be cycling between 70% and 78%. Abnormal Operating Procedure (AOP) 2B, Feedwater System Malfunction, was entered. An immediate in-plant inspection of main feedwater regulating valve 1CS-476B identified the positioner feedback arm was disconnected, with the connecting bolt nut located on the insulation covering the valve. Due to the inability to control steam generator level, operators initiated a manual reactor trip at 1517 hours and entered Emergency Operating Procedure (EOP) 0. An unexpected electrical lockout of 345 kV switchyard Bus Section 2 occurred. One moisture separator reheater drain valve failed to automatically shut and was manually shut. One turbine bearing oil lift pump failed to automatically start and was manually started. Reactor trip response was not negatively impacted by the unexpected conditions. After the reactor trip steam generator water levels were controlled using the feedwater regulating valve bypass valves. The plant was stabilized in Mode 3.

A root cause evaluation is in progress and initial indications are that less than adequate maintenance procedure guidance caused the loss of the positioner arm nut. Vendor guidance had not been included in maintenance procedures. Changes to the maintenance procedure will be tracked via the site's corrective action program. A safety function review has determined there was no safety impact from this event because the position controllers have no safety function.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

Event Description

On June 5, 2007, the Unit 1 reactor was manually tripped due to the inability to control main feedwater flow to the 'B' steam generator. The condition occurred because a nut for the bolt which secures the valve position feedback arm to the positioner output shaft came off allowing the two linkages to separate. The lack of position feedback caused control of the valve to be lost and steam generator level to cycle between 70% and 78%. A decision was made to manually trip the reactor. AOP 2B, Feedwater System Malfunction, was being utilized prior to the trip.

Unit 1 was in Mode 1 at 100% power prior to the reactor trip and was stabilized in Mode 3 following the trip. Motor-driven auxiliary feedwater pump P-38B and turbine-driven auxiliary feedwater pump 1P-29 appropriately started in response to plant conditions and were secured when steam generator levels were controlled using the feedwater regulating valve bypass valves.

Event Analysis

The event was a manual reactor trip in response to the inability to control level in one steam generator. Appropriate procedures were followed. Equipment and personnel functioned appropriately except for three minor unexpected conditions.

An unexpected electrical lockout of 345 kV switchyard Bus Section 2 occurred. One moisture separator reheater drain valve failed to automatically shut and was manually shut. One turbine bearing oil lift pump failed to automatically start and was manually started.

Bus Section 2 connects to the Unit 1 generator output. The lockout was in response to a delay in opening the A-phase pole. The delay was caused by improperly adjusted auxiliary switches in the breaker control cabinet. The improper adjustment was made during the previous refueling outage during a timing test performed by WE Energies personnel. The lockout of the bus section did not impact the response to the reactor trip. The preventive maintenance callup forms have been changed to include notification of System Engineers if auxiliary switches require adjustment.

One moisture separator reheater drain valve, 1FD-02603, failed to automatically shut and was manually, shut. The function of the drain valve is to prevent steam within the reheat system from reaching the turbine and causing an overspeed condition. The automatic failure did not impact the response to the reactor trip because a redundant valve, 1FD-02604, located in series with the affected valve, successfully shut, thus providing the same function.

One turbine bearing oil lift pump, 1P-129A for the Number 3 bearing, did not automatically start following the reactor trip. The pump was manually started. The cause of the start failure is being investigated via work order WO333160.

Steam generator 'B' level control was lost when the feedback arm from the main feedwater regulating valve became separated from the valve positioner. The separation was due to a nut coming off a connecting bolt. A root cause evaluation is being performed on the loss of the nut.

Cause

A manual reactor trip was initiated due to the loss of a nut with subsequent inability to control steam generator level. The cause for the loss of the nut has been determined to be a procedure inadequacy. Vendor technical bulletin information on the use of the specific type of locknut on positioner linkages was not completely incorporated into plant maintenance procedures.

Corrective Action

The corrective action, which will be tracked via the site's corrective action program, will be to revise the routine maintenance procedure to address linkage fasteners. Corrective Action (CA) 01095358-09 has been issued.

Safety Significance

The plant response during and following the transient, was as expected with three minor noted exceptions which did not impact any safety-related functions nor impair the ability to maintain the plant in a safe, controllable condition. Although the event was a manual actuation of the reactor protection system, plant equipment performance allowed a stable configuration to be maintained. Thus, the safety significance of the event was low and there was no impact on the health and safety of the public, and no impact on employee health and safety.

During this event and the subsequent recovery actions there was no loss of a safety-related system, structure or component. Therefore, the event did not involve a Safety System Functional Failure. The positioner components associated with the main feedwater regulating valves perform a function of normal steam generator water level control only. The safety function for the main feedwater regulating valves is to close on a main steam line break (MSLB). Solenoid valves, which are independent of the position controller, perform the safety function and were not affected by the loss of the connecting nut.

Component and System Description

The system affected was Feedwater/Condensate. The particular component was the main feedwater regulating valve; specifically the valve position control equipment. The positioner is a Bailey AP-4 equipped with a position feedback arm connected to a Copes-Vulcan Model 24 valve. The valve is a flow controlling, normally open valve in the main feedwater supply header for the Unit 1 'B' steam generator.

Previous Similar Events

A review of LERs submitted in the past three years, associated with requirement 10 CFR 50.73(a)(2)(iv)(A) for manual or automatic actuation of the reactor protection system, reactor trip, with a cause of procedural inadequacies, was conducted. No similar events were identified.

Failed Components Identified

None